Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

March 28, 2019

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Unit 1
Renewed Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Licensee Event Report 50-259/2018-002-01**

Reference: Letter from TVA to NRC, "Licensee Event Report 50-259/2018-002-00," dated May 17, 2018

On May 17, 2018, the Tennessee Valley Authority submitted Revision 0 to Licensee Event Report (LER) 50-259/2018-002-00 (Reference) which provided the details of an automatic reactor scram due to an unanticipated electro-hydraulic control logic condition. The enclosed LER has been revised to document a revised causal analysis, which determined that the event was due to a failure to identify operational risks.

The Tennessee Valley Authority is submitting the enclosed Licensee Event Report in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.73(a)(2)(iv)(A), as any event or condition that resulted in a manual or automatic actuation of the RPS and general containment isolation signals affecting containment isolation valves in more than one system or multiple Main Steam Isolation Valves.

There are no new regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact J. L. Paul, Nuclear Site Licensing Manager, at (256) 729-2636.

Respectfully,

D. L. Hughes
Site Vice President

Enclosure: Licensee Event Report 50-259/2018-002-01 – Automatic Reactor Scram Due to an Unanticipated Electro-Hydraulic Control Logic Condition

cc (w/ Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

NRC FORM 366 (04-2018) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001,or by e-mail to Infocollects. Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

1. Facility Name	2. Docket Number	3. Page
Browns Ferry Nuclear Plant, Unit 1	05000259	1 OF 6

4. Title

Automatic Reactor Scram Due to an Unanticipated Electro-Hydraulic Control Logic Condition

5. Event Date			6. LER Number			7. Report Date			8. Other Facilities Involved	
Month	Day	Year	Year	Sequential Number	Rev No.	Month	Day	Year	Facility Name	Docket Number
									N/A	N/A
03	18	2018	2018	002	01	03	28	2019	Facility Name N/A	Docket Number N/A

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. Power Level	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. Licensee Contact for this LER

Licensee Contact	Telephone Number (Include Area Code)
Ryan Coons, Licensing Engineer	256-729-2070

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES
D	JA	TD	EL22	Y	N/A	N/A	N/A	N/A	N/A

14. Supplemental Report Expected	15. Expected Submission Date	Month	Day	Year
☐ Yes (If yes, complete 15. Expected Submission Date) ☒ No		N/A	N/A	N/A

Abstract (Limit to 1400 spaces, i.e., approximately 14 single-spaced typewritten lines)

On March 18, 2018, at 1158 CDT, the Unit 1 reactor automatically scrammed due to a Reactor Protection System (RPS) signal generated from High Reactor Steam Dome Pressure in response to the Electro-Hydraulic Control (EHC) system clamping the Turbine Control Valves. All control rods fully inserted into the core. Main Steam Isolation Valves (MSIV) remained open with Main Steam Relief Valves operating on the initial transient as expected. Main Turbine Bypass Valves controlled reactor pressure. Reactor Feedwater pumps remained in service to control reactor water level. Primary Containment Isolation System Groups 2, 3, 6, and 8 containment isolation and initiation signals were received. Upon receipt of these signals all required components actuated as required. All safety systems operated as expected.

An investigation determined the event was caused by Operations and Engineering Management failing to track and review the aggregate impact of off-normal conditions. There was inadequate procedural guidance to assess operational impacts when equipment failures/faults occurred, prior to entering the work management process. The normal level of review would not have identified the condition leading to the scram.

Both megawatt input transducers were replaced prior to Unit 1 restart from the scram. As an interim action, a Nuclear Operating Experience Report (NOER) was issued for communication to other TVA sites. This NOER stressed the importance of evaluating the full impact of failed digital equipment, which must include software logic inputs. As a corrective action to prevent future recurrence, operator responsibilities for performing operational impact reviews were clarified in procedures. This included developing a tool or checklist for operational impact reviews.

NARRATIVE

I. Plant Operating Conditions Before the Event

At the time of discovery, Browns Ferry Nuclear Plant (BFN), Unit 1 was in Mode 1 at 100 percent power.

II. Description of Event

A. Event Summary

At 1158 CDT on March 18, 2018, the Unit 1 reactor automatically scrammed due to a Reactor Protection System (RPS) [JC] signal generated from High Reactor Steam Dome Pressure in response to the Electro-Hydraulic Control (EHC) [JA] system clamping the Turbine Control Valves (TCVs) [PCV].

All control rods [AA] fully inserted into the core. Main Steam Isolation Valves (MSIV) [ISV] remained open with Main Steam Relief Valves (MSRVs) [RV] operating on the initial transient as expected. Main Turbine Bypass Valves [V] controlled reactor pressure. Reactor Feedwater pumps [P] remained in service to control reactor water level.

Primary Containment Isolation System [JM] Groups 2, 3, 6, and 8 containment isolation and initiation signals were received. Upon receipt of these signals all required components actuated as required. All safety systems operated as expected.
The Tennessee Valley Authority is submitting this report in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.73(a)(2)(iv)(A), as any event or condition that resulted in a manual or automatic actuation of the RPS and general containment isolation signals affecting containment isolation valves in more than one system or multiple MSIVs.

B. Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event

The EHC system had no megawatt input logic, because both megawatt input transducers [TD] (Nexus modules) were non-functional and had been out of service since January 22, 2018.

NARRATIVE

C. Dates and approximate times of occurrences

Date	Time (CDT)	Event
April 8, 2017		EHC megawatt input transducer 2 became non-functional.
January 22, 2018		EHC megawatt input transducer 1 became non-functional.
March 18, 2018	1158	Unit 1 automatically scrammed due to a RPS signal generated from High Reactor Steam Dome Pressure in response to the EHC system clamping the TCVs.
March 18, 2018	1516	Operations personnel reported the scram to the NRC in Event Notification 53269.

D. Manufacturer and model number of each component that failed during the event

No components failed during this event.

E. Other systems or secondary functions affected

No other systems or secondary functions were affected by this event.

F. Method of discovery of each component or system failure or procedural error

This reactor scram event was identified through control room indicators and alarms.

G. Failure mode, mechanism, and effect of each failed component

No components failed during this event.

H. Operator actions

Appropriate operator actions were taken in response to the scram and recovery from the scram.

I. Automatically and manually initiated safety system responses

The RPS and containment isolation systems automatically responded due to this event, causing Unit 1 to scram and containment isolation.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Browns Ferry Nuclear Plant, Unit 1	05000259	2018	- 002	- 01

NARRATIVE

III. Cause of the Event

A. Cause of each component or system failure or personnel error

No systems or components failed during this event.

Operations and Engineering Management failed to track and review the aggregate impact of off-normal conditions.

B. Cause(s) and circumstances for each human performance related root cause

There was inadequate procedural guidance to assess operational impact at the time equipment failures/faults occurred, and prior to entering the work management process. The procedural guidance only assessed risk in the work management process which was specific to performing the work. There was no provision to assess the operational impacts of failed equipment before scheduling and planning repair work.

IV. Analysis of the Event

The scram was caused by having both U1 Nexus units failed and cycling the Generator 1 Relay 250V DC Supply Breaker (BFN-0-BKR-282-0024/709), during ground location activities. Opening the breaker dropped out relay coils (152Z1 and 152Z2), which each have a normally-open contact that provides Generator Breaker status to the EHC system. Closing the breaker re-energized these relay coils and closed the contact inputs to EHC. Transitioning the Generator Breaker status contacts from Open to Closed triggered the Initial Load Pickup logic, which takes the TCV Flow Demand signal (LSS Bus) and adds 4.7% (to prevent generator motoring) through a limiter (0-20%) and applies it to the LSS Bus. The TCV Flow Demand is then quickly ramped to the new position. In this instance, the LSS Bus was ramped from approximately 91% (which is approximately 59% TCV position) to 20% (which is approximately 10% TCV position). When the TCVs ramped to 10%, Bypass Valves all opened to 100%, but total flow steam flow was not adequate to prevent reactor pressure to raise subsequently resulting in an APR High Flux reactor scram.

The logic design was determined to be robust, in that it did not trip until there was an initiator (power cycling the generator breaker position relays) and two failures (the redundant MW devices). However, during review it was found that a loss of the only first stage shell pressure indicator would cause the same scram when cycling the breaker. The first stage shell pressure can be repaired/replaced while the unit is on-line.

On December 26, 2016, when one Nexus module failed, Operations challenged Engineering to identify the operational impact. The answer received was "The MVAR input is used for indication only. The MW input is used for indication only (except when using megawatt control which is normally only between speed control and pressure control during startup)". When the second Nexus module failed on January 22, 2018, Operations thought that the operational impact was minimal based on information previously provided. Identifying the operational impact of two failed Nexus

modules would have triggered actions to mitigate risk in accordance with NPG-SPP-07.3, Work Activity Risk Management Process.

The scram could have been prevented by taking risk mitigation actions when two Nexus modules had failed. The operational impact for this event was thought to be "none" based upon the system engineer's review of only one failed Nexus module. There was no official established process which drives detailed engineering reviews for aggregate equipment failures. The system engineer did not review the start-up logic because there was no guidance on engineering review depth.

In discussions with the EHC digital logic subject matter expert (SME), it was determined that it was unreasonable to expect the engineer to identify the risk during an operational impact review. These logic block diagrams are typically hundreds of pages long, and the engineer would not know which specific code or logic block diagrams to review. After the event, the EHC digital logic SME recognized the inputs and realized that the EHC logic caused the scram. The TCVs moving to 20% is the specific limit for control valves when the Generator Breaker is initially closed during starting up. Since the SME was involved in EHC code revisions, he was able to locate the logic in the logic block diagrams once this information had became available.

In hindsight, the risk assessments performed were inconsistent due to the depth of the aggregate operational impact review.

V. Assessment of Safety Consequences

This event automatically actuated safety systems and did not result in the inoperability or unavailability of any system to provide their required safety functions. All withdrawn control rods fully inserted into the core. Main Steam Isolation Valves (MSIVs) remained open with MSRVs operating during the initial transient as expected. The Main Turbine Bypass Valves controlled reactor pressure. Reactor Feedwater pumps remained in service to control reactor water level. Primary Containment Isolation Signals Groups 2, 3, 6, and 8 containment isolation and initiation signals were received. Upon receipt of these signals the affected components actuated as required. All safety systems operated as expected. Therefore, this condition was of low safety significance and had negligible impact on the health and safety of the public.

A. Availability of systems or components that could have performed the same function as the components and systems that failed during the event

No systems or components failed during this event.

B. For events that occurred when the reactor was shut down, availability of systems or components needed to shutdown the reactor and maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident

This event did not occur when the reactor was shutdown.

C. For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service

Safety system availability was not impacted by this event.

VI. Corrective Actions

This event was entered into the TVA Corrective Action Program and is being tracked under CR 1397341.

A. Immediate Corrective Actions

The immediate corrective actions included:

- Both megawatt input transducers were replaced prior to Unit 1 restart from the scram.
- Engineered Solutions, Inc. conducted an independent evaluation for what caused the Unit 1 scram.
- A Nuclear Operating Experience Report (NOER) was issued for communication to other TVA sites. This NOER stressed the importance of evaluating the full impact of failed digital equipment failure, which must include software logic inputs.

B. Corrective Actions to Prevent Recurrence or to reduce probability of similar events occurring in the future

Operator responsibilities for performing operational impact reviews were clarified in procedures. This included developing a tool or checklist for operational impact reviews.

VII. Previous Similar Events at the Same Site

A review of the BFN CAP and Licensee Event Reports for Units 1, 2, and 3 found no instances of reactor scrams similar to this event within the past five years.

VIII. Additional Information

None.

IX. Commitments

None.